Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

January 15, 2003
NOC-AE-03001452
File No.: G25
10CFR50.73
STI: 31542071

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Unit 1
Docket No. STN 50-498
Licensee Event Report 02-003
Manual Reactor Trip due to loss of cooling water

Pursuant to 10CFR50.73, South Texas Project submits the attached Unit 1 Licensee Event Report 02-003 regarding a manual reactor trip due to an apparent loss of open loop cooling water system pressure with subsequent auxiliary feedwater system actuation due to low steam generator water level.

This event did not have an adverse effect on the health and safety of the public.

Corrective actions 4, 5, 6, 7, 8, 9 and 10 are the only commitments contained in this LER

If there are any questions on this submittal, please contact W. R. Bealefield, Jr. at (361) 972-7696 or me at (361) 972-7849.

E. D. Halpin
Plant General Manager

Attachment: LER 02-003 (South Texas, Unit 1)

IE22
NRR

cc:

(paper copy)

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NRC FORM 366 (7-2001) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB NO. 3150-0104 | EXPIRES 7-31-2004

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

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4. TITLE
Manual reactor trip due to apparent loss of Open Loop Cooling Water System pressure

5. EVENT DATE MO	DAY	YEAR	6. LER NUMBER YEAR	SEQUENTIAL NUMBER	REV NO	7. REPORT DATE MO	DAY	YEAR	8. OTHER FACILITIES INVOLVED FACILITY NAME	DOCKET NUMBER
										05000
11	16	2002	2002	- 03 -	00	01	15	2003	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
William R. Bealefield, Jr.	361-972-7696

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE)	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On November 16, 2002 Unit 1 was operating at 100% power, at approximately 2036 the Unit 1 control room personnel received unexpected electrical system trouble alarms in the control room. Within several minutes Open Loop Cooling Water pump 12 tripped. The control room staff attempted to start Open Loop pump 13. Open Loop pump 13 did not start. The Reactor Operator (RO) noted that the Open Loop header pressure was approximately 30 to 40 psig and relatively stable. The yard watch operator that had been dispatched to the Circulating Water Intake Structure (CWIS) reported to the control room that the pump bay was filling with water and that it was believed to be coming from the Open Loop system header. While the off-normal procedure actions were being implemented for loss of Open Loop auxiliary cooling water, Open Loop header pressure indication in the control room dropped to zero psig. The Unit Supervisor directed the Unit 1 reactor be tripped manually and that Open Loop pump 11 be secured. Water level in the CWIS was observed to still be rising even after the last Open Loop pump was secured. Investigation revealed that the water was issuing from a large hole in the side of Circulating Water (CW) pump 11 casing. Circulating Water pump 11 was then secured.

An Auxiliary Feedwater (AFW) actuation occurred approximately 25 seconds following the manual reactor trip due to Lo-Lo water level in the 2C Steam Generator. The AFW system operated and maintained water level in all four Steam Generators.

All actuated safety related equipment operated as required.

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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

DESCRIPTION OF EVENT

On November 16, 2002 Unit 1 was operating at 100 % power. At approximately 2036 the Unit 1 control room personnel received unexpected electrical system trouble alarms in the control room. Within several minutes the Open Loop Cooling system pump 12 tripped. The control room staff entered the Loss of Open Loop Auxiliary Cooling Water procedure and attempted to start Open Loop pump 13. The Open Loop pump 13 should have started when Open Loop pump 12 tripped. Open Loop pump 13 did not start when operators attempted to start it. The Reactor Operator (RO) noted that Open Loop system header pressure was approximately 30 to 40 psig and relatively stable. The Yard Watch Operator that had been dispatched to the Circulating Water Intake Structure (CWIS) reported to the control room that the pump bay was filling with water and that it was believed to be coming from the Open Loop Auxiliary Cooling water system header. The Open Loop Cooling water system provides cooling for equipment essential for plant operation. While the off-normal procedure actions were being implemented, Open Loop header pressure indication in the control room dropped to zero psig. Based on these indications, the Unit Supervisor directed the Unit 1 reactor be tripped manually and that the Open Loop pump 11 be secured. The reactor was tripped and the Open Loop pump 11 was secured. The water level in the CWIS pump bay was observed to still be rising even after the last Open Loop pump was secured. Investigation by the yard operator revealed that the water was coming from a large hole in the side of Circulating Water (CW) pump 11. Circulating Water pump 11 was secured and the water level in the pump bay began to recede.

All actuated safety related equipment operated as required.
An Auxiliary Feedwater (AFW) actuation occurred approximately 25 seconds following the manual reactor trip due to Lo-Lo water level in the 2C steam generator. Based on emergency operating procedure guidelines and licensed operator simulator training, AFW actuation was not expected to occur until approximately 5 minutes later than it occurred.

Investigation revealed that the catastrophic failure of Circulating Water pump 11 pump casing was most likely caused by a pressure spike that occurred when the pump discharge valve actuator failed and allowed the valve to rapidly close. When the pump casing ruptured, the motor for Open Loop pump 12 shorted electrically and tripped. The Open Loop pump 13 seal water line was ruptured by flying debris which prevented this pump from starting. The Open Loop header pressure indication in the control room decreased to zero because a fuse blew for the indicating circuit when water entered a terminal box in the CW pump bay. The 96-inch discharge valves for the Circulating Water pumps are located close to the pump discharge, are butterfly type and are mounted in a vertical position. Having the discharge valves close to the pump discharge contributes to hydrodynamic instabilities in the discharge piping and valves which results in valve flutter which subjects the discharge valves and actuators to high stress and impacts. Any time the valve stem becomes separated from the valve actuator during pump operation, the valve will be rapidly closed by the very high water flow rate through the valve.
Failure scenarios similar to this event occurred in 1987 and in 1989. The reactor wasn't critical either time so no reactor trip occurred. On December 17, 1989, CW pump 11 had failed when the valve actuator spline adapter disengaged from the drive sleeve by moving down the stem of the discharge valve. The discharge valve closed and caused a pressure spike that resulted in the back of the pump casing being blown out. At that time, two designs were proposed to keep the spline adapter from disengaging. One design was for a clamp to be placed on the valve stem below the spline adapter to keep it from moving down the valve stem. The other design was a stiffback assembly that was

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attached to the top of the spline adapter and rested on top of the valve stem to prevent the spline adapter from dropping. Stiffback assemblies were installed on all the 96-inch valves in 1990. No stiffback assembly failures have occurred, prior to this event, since the original installations. The original stiffback assemblies were constructed of carbon steel and utilized carbon steel screws to attach the assembly to the spline adapter. The discharge valve actuator for CW pump 11 was replaced in May 2002 utilizing a new stiffback and associated machine screws. The stiffback that was installed was carbon steel but it was galvanized. The machine screws used were stainless steel vice carbon steel as specified in the original engineering change documents. The vendor manual specified that the countersunk screws be tightened snug tight and staked in place by upsetting the head of the screw into the stiffback plate. The staking on the screws in the actuator for CW pump 11 did not prevent the screws from rotating. Based on interviews with craft personnel, the consensus is that the staking marks on the edge of each of the screws were too small. The edges of the screw holes in the stiffback in this event were found with the screw holes wallowed out so any stake marks that were originally made in the stiffback have been worn away. Because the stiffback was galvanized carbon steel, the galvanized layer chipped when staked making it more difficult to verify that the stiffback side of the stake was adequate. During subsequent operation of CW pump 11 the spline adapter moved down the valve stem as the screws unthreaded from the adapter due to valve flutter impact. When the spline adapter disengaged from the drive sleeve, water flow rapidly closed the valve causing a pressure spike in the discharge portion of the pump.

Several CW system component failures and near misses have occurred that could cause a separation of the valve stem from its actuator. The consequences of this separation is usually that the valve will close which can damage other components due to the resultant pressure spike. There are weaknesses in the CW system design that contribute to component failures being caused by valve flutter induced impacts and vibration. Because the discharge valve is located closer to the pump than is recommended per engineering standards, the components within the valve and actuator are subjected to higher valve fluttering which causes impact and vibration stresses. The station has previously focused on the corrective actions for the individual failures and near misses but a comprehensive assessment of implications was not performed.

EVENT SIGNIFICANCE

This event is significant because it led to a reactor trip but there was not an unacceptable challenge to nuclear safety. The sequence of events and operator actions taken during this event are analogous to a turbine trip event. The conditional core damage probability given a turbine trip is 2.7E-07. Assuming one trip per year, the core damage frequency would be 2.7E-07 per year, which is below the Regulatory Guide 1.174 limit of 1E-06 per year for significant changes in core damage frequency.

CAUSE OF EVENT

Root Cause:

The root cause of this event is that the station failed to recognize the implications of the multiple material deficiencies and failure of components within the Circulating Water System.

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

Contributing Causes:

1. A contributing cause is that the impact of using alternate materials for the stiffback was not properly recognized and incorporated into the staking process. In addition, the engineering change document requirement for the screws to be carbon steel was not incorporated into the vendor manual. Because the stiffback was galvanized carbon steel, the galvanized layer chipped when staked making it more difficult to verify that the stiffback side of the stake was adequate. Because the screws were stainless steel vice the carbon steel listed in the bill of materials in the engineering change documents, more force would have been required to obtain a good stake than would be required with carbon steel.

2. A contributing cause is that the staking on the stiffback machine screws was not verified to have been performed in accordance with the vendor manual instructions which required that the countersunk screws be staked by upsetting the head of the screw into the stiffback plate.

CORRECTIVE ACTIONS

1. Unit 1 Circulating Water system was aligned to isolate CW pump 11. The CW system was returned to service and Unit 1 was returned to service.
   Complete

2. Short-term improvements have been implemented for the Unit 1 CW pumps 12, 13, 14 and Unit 2 CW pumps 21, 22, 23, 24. A clamp was installed in each pump's discharge valve a short distance below the spline adapter to serve as a backup in case the stiffback does not adequately restrain the adapter.
   Complete

3. A Condition Report Engineering Evaluation was prepared for the Delta 94 steam generator response to the reactor trip explaining why the Auxiliary Feedwater system actuated earlier than expected and defining the expected response in the future.
   Complete

4. A case study will be developed and presented as recommended in Significant Operating Experience Report (SOER) 02-4, "Reactor Pressure Vessel Head Degradation at Davis-Besse Nuclear Power Station".
   The case study will include the organizational factors from this event with an emphasis on understanding the relative consequences of degraded plant conditions, standards for station equipment performance, and the expected response of aggressively following up and resolving degraded plant conditions.
   This corrective action will be completed by March 31, 2003.

5. The station will perform a Graded Quality Assurance (GQA) and Plant Generation Risk (PGR) ranking of the CW system functions and components.
   This corrective action will be completed by March 27, 2003.

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6. The station will rescope all CW system functions and components that get GQA or PGR risk ranked as high or medium in the Maintenance Rule and appropriate performance criteria will be established.
   This corrective action will be completed by June 5, 2003.

7. The vendor manual for the CW pump discharge valves will be revised to incorporate the requirement for the stiffback machine screws to be made of carbon steel.
   This corrective action will be completed by February 27, 2003.

8. Engineering will perform a detailed analysis of the circulating water system to identify additional subcomponents that are susceptible to failure due to valve flutter and implement actions to prevent failures. This will include an evaluation of the effectiveness of the new style valve disc installed in CW pump 21 discharge valve in reducing valve flutter as well as other possible corrective actions.
   This corrective action will be completed by June 5, 2003.

9. The training department will evaluate the adequacy of training on "staking" including methods for verifying that the staking is adequate.
   This corrective action will be completed by February 27, 2003.

10. Maintenance department personnel will be briefed on the fact that existing station documentation does not allow substitution of materials that are "as good or better" as the specified materials unless properly authorized. This corrective action will completed by March 1, 2003.

ADDITIONAL INFORMATION

Circulating Water pump 11 casing is scheduled to be replaced during the next Unit 1 refueling outage.

AFW was initiated, during this event, by a Lo-Lo level signal for the 2C Steam Generator (S/G) at a level of 20% approximately 25 seconds after the manual trip of Unit 1 from 100% power. Based on Licensed Operator training in the simulator, this actuation was anticipated to occur approximately 5 minutes later than its actual occurrence. Integrated Computer System data of S/G level versus time throughout the transient shows that S/G's A and C dropped to minimum levels of approximately 19.8% and 17.5% respectively while S/G's B and D dropped to minimum levels of approximately 20.6% and 22.7%.

The steam generators installed in Units 1 and 2 are Westinghouse Model Delta-94. Section 2.2T of the Delta-94 Thermal and Hydraulic Design and Data Report provides a graph of anticipated S/G levels following a reactor trip for power levels from 0%-100%. At 100% power, the graph indicates that the hot standby water level is less than 3 inches from the S/G Lo-Lo setpoint. The graph was developed using a steady-state model with simplified and symmetrical assumptions. The 1.4% power uprate was not factored into the graph. When adjustments were made for additional feedwater flow post-trip (approximately 13 seconds), possible steam flow asymmetries, individual S/G power outputs and instrument uncertainties, the response of the Delta-94 S/Gs post-trip was consistent with the design basis analyses.

Following engineering evaluation, recommendations will be provided on steam generator operating levels and setpoints such that S/G Lo-Lo level actuations are precluded.